UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated November 30, 1995, as supplemented on March 15, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Facility Operating License No. DPR-69 is hereby amended to read as follows:

C. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 190, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Susan Frant Shankman, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 22, 1996

<u>ATTACHMENT TO LICENSE AMENDMENTS</u>

<u>AMENDMENT NO. 213 FACILITY OPERATING LICENSE NO. DPR-53</u>

<u>AMENDMENT NO. 190 FACILITY OPERATING LICENSE NO. DPR-69</u>

<u>DOCKET NOS. 50-317 AND 50-318</u>

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-10	3/4 4-10
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-16	3/4 4-16
B3/4 4-4	B3/4 4-4
B3/4 4-5	B3/4 4-5
B3/4 4-6	B3/4 4-6*
B3/4 4-7	B3/4 4-7*

*Indicates rollover pages.

3/4.4 **REACTOR COOLANT SYSTEM**

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be **OPERABLE**.

APPLICABILITY: **MODES** 1, 2, 3 and 4.

ACTION: With one or more steam generators inoperable, restore the inoperable generator(s) to **OPERABLE** status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined **OPERABLE** during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (> 20%), and

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

7. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

   a. original tube wall..................................40%
   b. Westinghouse laser welded sleeve wall.............40%

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

SURVEILLANCE REQUIREMENTS (Continued)

10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following process:

    a) Westinghouse Laser Welded Sleeving as described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

    Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per Specification 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

b. The steam generator shall be determined **OPERABLE** after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 15 days pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed (pursuant to Specification 6.9.1.5.b). This report shall include:

    1. Number and extent of tubes inspected.

    2. Location and percent of wall-thickness penetration for each indication of an imperfection.

    3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days pursuant to Specification 6.9.2.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per SG.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this SG, plug or repair defective tubes and inspect 2S tubes in each other SG. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	All other SGs are C-1	None	N/A	N/A
			Some SGs C-2 but no additional SG are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug or repair defective tubes. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspections.

2. An assessment of the structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," that can be expected before the end of the fuel cycle or 30 months, whichever comes first.

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon perminute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired. Defective tubes may be repaired by a Westinghouse Laser Welded Sleeve. The technical bases for Westinghouse Laser Welded Sleeve are described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections at or exceeding the plugging or repair limit of 40% of the original tube nominal wall thickness. If a tube contains a Westinghouse Laser Welded Sleeve with imperfection exceeding 40% of nominal wall thickness, it must be plugged. The basis for the sleeve plugging limit is based on Regulatory Guide 1.121 analyses, and is described in the Westinghouse sleeving technical report mentioned above. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Repaired tubes are also included in the inservice tube inspection program.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specifications 6.9.2 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 Reactor Coolant System Leakage

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM **IDENTIFIED LEAKAGE** limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of **UNIDENTIFIED LEAKAGE** by the Leakage Detection Systems.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 100 gallon per day leakage limit per steam generator ensures that steam generator tube integrity is maintained in accordance with the recommendations of Generic Letter 91-04.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any **PRESSURE BOUNDARY LEAKAGE** requires the unit to be promptly placed in **COLD SHUTDOWN**.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion.

Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the **SITE BOUNDARY** will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Calvert Cliffs site, such as **SITE BOUNDARY** location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The **ACTION** statement permitting **POWER OPERATION** to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram **DOSE EQUIVALENT I-131**, but within the allowable limit shown on Figure 3.4.8-1, accommodates possible iodine spiking phenomenon which may occur following changes in **THERMAL POWER**. Operation with specific activity levels exceeding 1.0 μCi/gram **DOSE EQUIVALENT I-131** but within the limits shown on Figure 3.4.8-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4.8-1 increase the 2 hour thyroid dose at the **SITE BOUNDARY** by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and **STARTUP** and shutdown operation. The various categories of load cycles used for design purposes are provided in Section 4.1.1 of the UFSAR. During **STARTUP** and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Operation within the appropriate heatup and cooldown curves assures the integrity of the reactor vessel against fracture induced by combinative thermal and pressure stresses. As the vessel is subjected to increasing fluence, the toughness of the limiting material continues to decline, and ever more restrictive Pressure/Temperature limits must be observed. The current limits, Figures 3.4.9-1 and 3.4.9-2, are for a peak neutron fluence to the inner surface of the reactor vessel of $\leq 2.61\times10^{19}N/cm^2$ ($E > 1$ MeV). This fluence corresponds to the Pressurized Thermal Shock Screening Criteria defined in 10 CFR 50.61 for weld 2-203 A, B, C.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Section 4.1.5 of the UFSAR. Reactor operation and resultant fast neutron ($E > 1$ MeV) irradiation will cause an increase in the RT_{NDT}. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4-13 and are approved by the NRC prior to implementation in compliance with the requirements of 10 CFR Part 50, Appendix H.

The shift in the material fracture toughness, as represented by RT_{NDT}, is calculated using Regulatory Guide 1.99, Revision 2. For a fluence of $2.61\times10^{19}N/cm^2$, the adjusted reference temperature (ART) value at the 1/4 T position is 241.4°F. At the 3/4 T position the ART value is 181.0°F.

These values are used with procedures developed in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G to calculate heatup and cooldown limits in accordance with the requirements of 10 CFR Part 50, Appendix G.

To develop composite pressure-temperature limits for the heatup transient, the isothermal, 1/4 T heatup, and 3/4 T heatup pressure-temperature limits are compared for a given thermal rate. Then the most restrictive pressure-temperature limits are combined over the complete temperature interval resulting in a composite limit curve for the reactor vessel beltline for

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be **OPERABLE**.

APPLICABILITY: **MODES** 1, 2, 3 and 4.

ACTION: With one or more steam generators inoperable, restore the inoperable generator(s) to **OPERABLE** status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined **OPERABLE** during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

    1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

7. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

   a. original tube wall...................................40%
   b. Westinghouse laser welded sleeve wall.............40%

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following process:

    a) Westinghouse Laser Welded Sleeving as described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

    Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per Specification 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

b. The steam generator shall be determined **OPERABLE** after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 15 days pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed (pursuant to Specification 6.9.1.5.b). This report shall include:

    1. Number and extent of tubes inspected.

    2. Location and percent of wall-thickness penetration for each indication of an imperfection.

    3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days pursuant to Specification 6.9.2.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per SG.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this SG, plug or repair defective tubes and inspect 2S tubes in each other SG. 24 hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	All other SGs are C-1	None	N/A	N/A
			Some SGs C-2 but no additional SG are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug or repair defective tubes. 24-hour verbal notification to NRC with written followup pursuant to Specification 6.9.2.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspections.

2. An assessment of the structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," that can be expected before the end of the fuel cycle or 30 months, whichever comes first.

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired. Defective tubes may be repaired by a Westinghouse Laser Welded Sleeve. The technical bases for Westinghouse Laser Welded Sleeve are described in the proprietary Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections at or exceeding the plugging or repair limit of 40% of the tube original nominal wall thickness. If a tube contains a Westinghouse Laser Welded Sleeve with imperfection exceeding 40% nominal wall thickness, it must be plugged. The basis for the sleeve plugging limit is based on Regulatory Guide 1.121 analyses, and is described in the Westinghouse sleeving technical report mentioned above. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Repaired tubes are also included in the inservice tube inspection program.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specifications 6.9.2 prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 Reactor Coolant System Leakage

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM **IDENTIFIED LEAKAGE** limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of **UNIDENTIFIED LEAKAGE** by the Leakage Detection Systems.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 100 gallon-per-day leakage limit per steam generator ensures that steam generator tube integrity is maintained in accordance with the recommendations of Generic Letter 91-04.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any **PRESSURE BOUNDARY LEAKAGE** requires the unit to be promptly placed in **COLD SHUTDOWN**.

3/4.4.7 CHEMISTRY

The limitations on RCS chemistry ensure that corrosion of the RCS is minimized and reduce the potential for RCS leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the RCS over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the **SITE BOUNDARY** will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Calvert Cliffs site, such as **SITE BOUNDARY** location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The **ACTION** statement permitting **POWER OPERATION** to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram **DOSE EQUIVALENT I-131**, but within the allowable limit shown on Figure 3.4.8-1, accommodates possible iodine spiking phenomenon which may occur following changes in **THERMAL POWER**. Operation with specific activity levels exceeding 1.0 μCi/gram **DOSE EQUIVALENT I-131** but within the limits shown on Figure 3.4.8-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4.8-1 increase the 2 hour thyroid dose at the **SITE BOUNDARY** by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the RCS are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and **STARTUP** and shutdown operation. The various categories of load cycles used for design purposes are provided in Section 4.1.1 of the UFSAR. During **STARTUP** and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Operation within the appropriate heatup and cooldown curves assures the integrity of the reactor vessel against fracture induced by combinative thermal and pressure stresses. As the vessel is subjected to increasing fluence, the toughness of the limiting material continues to decline, and even more restrictive Pressure/Temperature limits must be observed. The current limits, Figures 3.4.9-1 and 3.4.9-2, are for up to and including a fluence of 4.0×10^{19} n/cm^2 (E > 1 Mev) at the clad/vessel interface, which corresponds to approximately 30 Effective Full Power Years.

The reactor vessel materials have been tested to determine their initial RT_{NDTv}; the results of these tests are shown in Section 4.1.5 of the UFSAR. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4-13 and are approved by the NRC prior to implementation in compliance with the requirements of 10 CFR Part 50, Appendix H.

The shift in the material fracture toughness, as represented by RT_{NDT}, is calculated using Regulatory Guide 1.99, Revision 2. For a fluence of 4.0×10^{19} n/cm^2, the adjusted reference temperature (ART) value at the 1/4 T position is 177.1°F. At the 3/4 T position the ART value is 146.8°F. These values are used with procedures developed in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G to calculate heatup and cooldown limits in accordance with the requirements of 10 CFR Part 50, Appendix G.

To develop composite pressure-temperature limits for the heatup transient, the isothermal, 1/2 T heatup, and 3/4 T heatup pressure-temperature limits are compared for a given thermal rate. Then the most restrictive pressure-temperature limits are combined over the complete temperature interval resulting in a composite limit curve for the reactor vessel beltline for the heatup event. The composite pressure-temperature limit for the